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Abstract

Field Observations of Environmental Exposure dose from Radio-Argon Clouds

S. Sakagishi, T. Ishihara and I. Miyanaga

Health Physics Division

Japan Atomic Energy Research Institute

^{41}Ar produced in the air found in irradiating pockets of beam tubes is the only gaseous radioactive material produced during normal operation of certain types of reactors that can cause external exposure hazards to neighboring population. At the same time, this provides a good and effective tracer to determine experimentally diffusion patterns of gaseous waste into atmosphere.

In the Health Physics Division of the Japan Atomic Energy Research Institute field observations were undertaken to measure ^{41}Ar for the dual purpose of determining the environmental exposure dose by ^{41}Ar from several reactors, and for checking experimentally the dose estimation by Pasquill-Meade's method, which has been adopted by the Atomic Energy Commission of the Japanese Government as standard method of analysis for diffusion problems.

The observations were carried out 4 times altogether in the period from 1961 to 1962, using JRR-2 which is an enriched uranium fueled heavy water reactor of 10 MW.

The results of observation showed that the estimation of external dose by Pasquill-Meade's method agreed with measured data in first approximation, and it was concluded that it is satisfactory to use the

method to solve problems in practice. The last run of observations was carried out in so called "calm condition". In this case, the dose distribution curves were found to be quite complex and it was difficult to analyse the results completely; but the obtained values give an indication of the correction factor to be applied to calculated values for a safe-side estimation of the dose.

In the paper, a description is given of the field observations undertaken, including the instruments used, together with meteorological data and isodose patterns in each run of observation, as well as some analysis on population dose based thereon.

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Abstract

THE PERIODIC SYSTEM OF ATOMIC NUCLEI AND THE POSSIBILITY OF NEW ISOTOPE APPLICATION IN SCIENCE AND TECHNOLOGY

I.P.Selinov

I. The periodic system of atomic nuclei.

The rational systematization of nuclids reflects the regularities of atom nucleus structure, systemizes the isotope properties and facilitates the new isotope search.

The determination of a number of periodical regularities in masses, mass numbers, abundance and other properties of isotopes makes it possible plotting of atomic nucleus periodic system(on the bases of shell model of nucleus).

The paper desribes the gelionic type of building 2β-stable nuclids and the analogic isotope concept.

The proposed atomic nuclei system is described. The principal identity of different forms of the nuclid system which depicture the same regularities in atomic nuclei plotting is shown.

II. The possibility of new isotope discovery and application in science and technology.

Stable isotope mass numbers are predicted in the old variant of the authors's periodic system of isotopes.

Radioactive nuclids are systemized and mass numbers of -stable isotopes of transuranium elements are predicted . The regularities in half-lives comprise the basis of search for new isotopes

The possibility of discovery of new isotopes useful as radiotracers (Ti^{52} etc) and in geological age determination is shown. The possibility of use of some new fissible isotopes as compact neutron sources particularly for activation analysis is described.

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The probability of symmetric fission of ekapolonium -stable isotopes (Z=106) and the possibility of existence of neutronless nuclid sources of nuclear energy very useful for practical purposes are p reported.

The anomalies in abunances of lantanide isotopes are explained by cosmic occurence of these isotopes due to the fission of superheavy nuclids with atomic numbers Z=106, 116 and with magic numbers of neutrons. The increase of half-lives of a number isotopes of these translawrencium elements and the possibilities of their artificial production are proposed.

III. The organization of regular and quick publication of international reference tables of isotopes and nuclear constants is described as purposeful.

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Abstract

PHYSICAL START-UP OF THE REACTOR AT THE BELOYARSK ATOMIC POWER STATION NAMED AFTER I.V.KURCHATOV

L.V.Konstantinov, B.V.Orlov,
B.V.Bondarenko et al.

The paper describes the results of the experiments which were carried out to study the physical characteristics of the reactor at the Beloyarsk atomic power station named after I.V.Kurchatov. The physical start-up of the reactor was carried out is September-December of 1963. The first criticality was achieved on September 3,1963.

At the beginning of the experiments several critical assemblies with a uniformly enriched uranium lattice were assembled and studied in the graphite stacking or the reactor; a study was also made of assemblies with a non-uniform lattice formed by the fuel elements of two types differing in the degree of uranium enrichment.

The study of each assembly included the measurement of the neutron flux distribution along the radius and height of the reactor and determination of the reactivity effect from the fuel channels, from the channels of the safety and control rods and from other disturbances.

Resonance absorption in uranium-238 was measured with the help of experimental channels specially made for the purpose.

After the assemblies had been studied with a minimum critical mass the reactor was fully loaded by the fuel channels. With the operating charge of the reactor a study was made of the effect produced on the reactivity by the water in the fuel channels.

These experiments have confirmed that the reactivity changes but little when water is blown away from the fuel channels, which is especially important for stable performance of the reactor at

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the beginning of power operation when the density of the coolant sharply changes.

When making investigations with the operating charge considerable attention was devoted to the study of the neutron flux distribution within the reactor and the dependence of this distribution on various positions of the absorbing rods in the reactivity compensation system.

The reactivity margin, the intensity of the temperature effect and of the poisoning effect were determined by the changes in the physical "weight" of the absorbing rods.

Reactor monitoring during the physical start-up as well as the experimental work were carried out with the help of special highly sensitive instruments (neutron flux indicators, reactivity meters, harmonic master analyser for the measurement of the neutron flux distribution and of the weights of the absorbing rods by their oscillation). Some of these instruments will be used for the investigation of the transiensts in the reactor and for determining its frequency characteristics.

The experiments and calculations carried out during the physical start-up have made it possible to establish the main physical characteristics of the reactor and to suggest important recommendations in respect to the operational control of the reactor.

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Abstract

PHYSICAL INVESTIGATION OF WWPR REACTOR AT NOVO-VORONEZH ATOMIC POWER STATION

L.V.Komissarov, G.L.Lunin, A.N.Novikov, V.A.Sidorenko

The paper describes WWPR of the Novo-Voronezh atomic power station and gives the physical characteristics of its core. It discusses the main physical problems involved in the development of such reactors, the interdependance of physical and operational characteristics of the reactor, particularly, the problems of nuclear safety peculiar to a large water-cooled water-moderated reactor which has a comparatively small number of effective control organs. Attention is drawn to the problem of "local criticality".

The paper summarized the results of investigations with respect to various physical characteristics of the core in a water reactor employing critical assemblies and discusses the problems connected with the calculation analysis of the cores in such reactors and refinement of the calculation techniques on the basis of the experimental data. The main emphasis is placed on the investigation of the efficiency of various control components, their interaction in the core, deformation of the neutron field and of the heat-release field.

This is followed by the calculation analysis of the core in WWPR of the Novo-Voronezh atomic power plant and by comparison between the design and experimental characterisitcs obtained from the investigation of the operating core of the reactor.

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Abstract

URANIUM-WATER CRITICAL ASSEMBLIES

N.A.Lazukov, A.Y.Gagarinsky, S.I, Garanin, G.V.Shyshkin

The paper summarises many years of experience gained in the course of operation on the test devices for uranium-water critical assemblies. It defines the basic requirements to the assemblies which, if complied with, make it possible to quickly simulate various types of reactors and describes the design of a typical tank-type and high-temperature test installations and the monitoring, control and safety system.

The basic provisions for the nuclear safety on critical assemblies are described along with the operating procedure for these assemblies.

The paper sets forth the organizational principles of radiation monitoring and radiation safety systems, describes the set of monitoring devices for test installations and individual monitoring devices and cites the average figures on personnel irradiation for the last several years of operation.

The paper then discusses the measuring accuracy and the methods of measurement of main neutron-physical parameters of uranium-water lattices: reactivity, breeding ratio, material buckling, migration area, etc.

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Abstract

THE THERMAL METHOD OF VITRIFICATION OF THE RADIOACTIVE PULPS AND THE SAFE DISPOSAL PROBLEM OF VITREOUS PREPARATIONS

Zimakov P.V. et al.

The present report discusses the problem of a continuous thermal treatment process for radioactive pulps containing the metal hydroxide mixture into vitreous preparations and the conditions of safe disposal of these preparations. The report describes the process physico-chemical principles and bases the use of a "fluidized bed" of quartz sand for dehydrati on of the above mentioned pulps and calcination of their solid residue.

The principal plant flow diagram is described and the main results of its testing on the pilot plant are given. The pilot plant consisted of the following components.

1. The high-efficiency apparatus with a "fluidized bed"
2. The continuous melting apparatus fitted with proportioning devices for slurry feeding and the bottom feeder for loading the melt into containers.

The studying of the radiation stability of vitreous preparations has shown that surface destruction occurred during the irradiation. It leads to the increased solubility of the preparations and to the formation of the mechanically undurable and radiation instable products on the surrace.

It was found that the destruction of the surface was caused by the radiation chemical reactions of the components of fused preparations with the air components.

The process rate decreases on increasing the temperature as

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the gas adsorption on the surface of the preparations decreases and the annealing of radiation defects takes place.

The more simple storage constructions for the high-active vitreous preparations as compared to the storages for the liquid and the other types of solid wastes have been proposed on the basis of complex investigations of properties of vitreous preparations.

The optimum temperature of ground disposal about 700°K during the first period of storage is recommended on the basis of experiments with real vitreous preparations (sp.ac. 2-10 10^3 Cu/l) and electrical modelling with simulated preparations as well as on the basis of thermophysical calculations.

The storage dimensions and arrangement are specially defined.

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Abstract

THE NERO-REACTOR FOR SHIP-PROPULSION

M. Bogaardt and M. Muysken
Reactor Centrum Nederland, The Hague, the Netherlands

This paper describes the design of the NERO-reactor for ship-propulsion and the experimental work performed to verify the design parameters of the reactor and auxiliary equipment.

The design covers a pressurized water reactor installation in which special consideration has been given to obtain a compact installation, a high degree of reliability and safety and a long core life. Novel features of the design are inter alia the proposed use of burnable poison incorporated in the uranium-dioxide fuel, an increased water circulation through the core as compared with the circulation through the external circuit obtained by hydraulic ejectors inside the reactor vessel, and a combined unit for heat exchanger, superheater and pump for the external circuit.

The design is being established and the experimental work is being performed under an association contract between R.C.N. and Euratom, effective since June 1961. The experimental work includes the performance of critical experiments with the KRITO-installation at Petten, the preparation of fuel pellets incorporating burnable poison, performance testing of fuel rods under irradiation in the High Flux Reactor at Petten, dynamic measurements on a model heat exchanger, for which an installation is being erected at the Technological University of Eindhoven, and on a full sized pressurizer in operation at the Technological University of Delft.

Contracts have been given to industry for development of several items, and investigations concerning corrosion, welding problems, heat extraction, hydraulics and shielding are under way.

The paper will include the results of experimental work obtained thus far.

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Abstract

HEAT TRANSFER AND STABILITY IN BOILING WATER REACTORS

M. Bogaardt, F.J.M. Dijkman and C.L. Spigt
Technological University, Eindhoven, the Netherlands

Fundamental work on heat removal by bubble formation and by increased heat diffusivity as a result of increased turbulence in boundary layers is being done in this laboratory.

A mathematical model has been constructed of the heat transfer loop that is in operation under pressures up to 40 atm., at input powers up to 600 kW and inlet subcooling up to 40°C. The model is compared with the experimental results obtained from the loop.

The on-set of flow instability is studied and the nature of the instabilities is characterised in the region up to burn-out by means of analysis of power density curves. Again, pressure and inlet subcooling are varied.

The dynamic characteristics of the loop are studied by means of transfer function alaysis. The validity of the underlying linearisation is discussed and the resulting transfer functions are compared with those obtained from noise analysis.

Finally, systematic burn-out studies of rod bundles are reported, including the analysis of flow instability preceding burn-out.

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Abstract

Burn-out in subcooled forced convection boiling of polyphenyls

D.A. van Meel (session 1.9-a)
Central Technical Institute T.N.O., The Hague, the Netherlands

Since 1961 T.N.O. has been carrying out an extensive research program on the influence of liquid velocity, degree of subcooling and liquid composition on the critical heat flux of polyphenyls flowing upward through a round stainless steel tube heated by the Joule effect.

A description of the test loop is given.

Results are reported on a great number of runs with terphenyl mixtures pressurized by direct contact with nitrogen. These results are well correlated by an equation of the type:

$$\Phi = a + b \cdot G + c \cdot G^{\frac{3}{4}} \cdot \Delta t_s$$

(Φ, critical heat flux; G, mass velocity; Δt_s, degree of subcooling).

The values of the coefficients will be given for the following compositions:
diphenyl 1 wt.%, o-terphenyl 14-16 wt.%, m-terphenyl 79-81 wt.%, p-terphenyl 3-5 wt.%.
diphenyl 1 wt.%, o-terphenyl 64-66 wt.%, m-terpenyl 31-33 wt.%, p-terphenyl 2-4 wt.%.

Parameters as for instance the degree of subcooling are based on the conditions at the outlet of the test tube, i.e. close to the location of burn-out.

The heat flux at the moment at which a sharp increase of the resistance of the test tube is indicated by a burn-out detector is interpreted as the real burn-out heat flux. Some tests in which the test tube was damaged by effecting real burn-out showed this to be permissible.

After these measurements on terphenyls under nitrogen,

the loop has been modified in such a way that liquid and pressurizing medium are separated from each other in a diving bell by molten solder. This enables the study of the separated effects of low-boilers and gases.

Results are presented on terphenyls containing no volatile components and on the effects of increasing concentrations of nitrogen, benzene and diphenyl. These results are interpreted on the basis of the degree of subcooling as derived from the experimentally determined vapour pressure curves. For this purpose a vapour pressure apparatus was developed permitting the accurate measurements of vapour pressure as a function of temperature and composition of mixtures of terphenyls and very volatile substances. This apparatus, which is connected permanently to the test loop, is characterized by the possibility of measuring under the conditions of a very small vapour space, as a consequence of which changes of the liquid composition are avoided.

To obtain equilibrium conditions the liquid may be stirred during the measurement.

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Abstract

IMPROVED ALUMINIUM ALLOYS AS FUEL CLADDING MATERIAL IN WATER COOLED POWER REACTORS

by

Steinar Aas and Ketil Videm
Institutt for Atomenergi
Kjeller, Norway

Aluminium cladding material can reduce the fuel cost in a boiling heavy water reactor by 0.1 to 0.4 mills/kWh depending upon wall thickness, enrichment and other factors.

Extensive loop testing of aluminium alloys of nominal composition Al-10 $^{w}/_{o}$ Si, 1 $^{w}/_{o}$ Mg and 1 $^{w}/_{o}$ Ni at 265°C and 315°C has shown that the alloys have a general corrosion rate of 0.03 mm/year. This low rate makes these alloys readily applicable in water cooled power reactors. Creep tests at elevated temperatures show that the 10% Si, 1% Mg- alloys have appreciably better mechanical properties than 2 S or X8001. The creep strength is nevertheless not good enough to make the cladding-tube freestanding. The metal will consequently contract and expand according to the movement of the fuel. These strains may lead to fatigue failure and results from strain-fatigue tests are given. These show that the alloy will have a satisfactory life time.

The cladding tube may also fail because of ratcheting or similar phenomena and these problems are reviewed.

The assembly of fuel stringers with long non-free standing cladding tubes presents some difficulties and some experimental designs are shown. One of these is presently being tested at 315°C in an out-pile-loop. Elements of a similar design are being made for testing in the Halden Boiling Water Reactor in the very near future.

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Abstract

MILD STEEL IN PRIMARY CIRCUITS OF WATER COOLED POWER REACTORS

by

Ketil Videm
Institutt for Atomenergi
Kjeller, Norway

Evaluation of the possibilities for substituting stainless steels with normal boiler steels in a water cooled power reactor requires knowledge of the quality of the water and the steam in every part of the system and the behaviour of the steel in these environments. The present survey deals mainly with corrosion and purification problems in a mild steel system and the considerations are based on a direct cycle boiling water reactor.

The steam from a boiling water reactor will normally contain from 5 - 20 ppm oxygen. It is possible to reduce the oxygen content by injecting hydrogen in the reactor. It is also possible to raise the pH to values used in conventional plants by adding ammonia. However, results from experiments carried out in HBWR (covered in a special paper at this conference) show that a very high dosage of hydrogen is required in both cases. It would be a more attractive solution if mild steel could be applied in neutral water with radiolytic oxygen and hydrogen present.

A series of corrosion experiments in out-of-pile loops both at high and low temperatures have been carried out to this end, and the results are compared with litterature data and with corrosion experiments carried out in a boiling water power reactor (HBWR). Special effort has been devoted to the study of the effect of the oxygen content and pH, and of methods for preconditioning of the steels in order to minimize the corrosion and corrosion product release.

It is concluded that the high temperature corrosion resistance of normal boiler steel in boiling water reactor environment is acceptable and that corrosion of components at low temperature and attack during shut down on parts normally at high temperatures are the problems most difficult to handle.

In all circumstances the application of carbon steel or low alloy steels will result in more corrosion products being given off to the water and a more effective purification system is required.

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Abstract

THIN WALLED STAINLESS STEEL AS FUEL CLADDING MATERIAL IN WATER-COOLED POWER REACTORS.

by

E. Brodahl, R.H. Søvik, K. Videm and S. Aas
Institutt for Atomenergi
Kjeller, Norway

ABSTRACT

The relatively high neutron absorption cross section of stainless steel indicates that an appreciably improved economy can be achieved through reduction of the wall thickness. Reducing the wall thickness means increased stresses and the tube will eventually collaps on to the fuel which will carry the outer pressure. Results from investigations of the mode of collaps as function of tube-diameter, wallthickness and gap are reported. The fission gas pressure may in certain cases become large enough to deform the canning. Results from experiments in which calculated and observed yield- and bursting pressures have been correlated are given.

The general corrosion rate of austenitic stainless steel is considered very satisfactory in water. Pitting corrosion or other forms of non-uniform corrosion attack is more dangerous as pits may very easily penetrate the thin wall. Observations from some experiments into this matter are given. Welding of thin-walled tubes requires careful plug design and welding technique. Examples of plug-designs, welding cycles and equipment are given. The assembly of long thin-walled fuel stringers into clusters can be done by brazing. Equipment and procedures are described. Assembly can be done by using wire spacers and some experiences with such types are reported.

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Abstract

OPERATIONAL EXPERIENCE ON THE 2ND CHARGE FOR THE HALDEN BOILING HEAVY WATER REACTOR

by

E. Jamne, J.A. Firing, M. Øvreeide and T. Wullum[+)]

Institutt for Atomenergi

OECD Halden Reactor Project

Halden, Norway

After termination of the 1st fuel charge operation in April 1961 the Halden Boiling Heavy Water Reactor was extensively modified for the 2nd charge operation. The reactor was taken critical in March 1962. After series of low power experiments, the initial approach to power was made in October 1962. Up to the end of 1963 the plant has been operated for experimental purposes on variable power levels up to the design power (20 M W). Most of this operation has been at 230°C or 28.5 ato reactor pressure.

Operation has been interrupted at intervals due to changes in the experimental set up and partly due to maintenance and modification. As could be expected in an experimental operation of this kind, involving so much experimental equipment, there has been a number of automatic shut down actions. The reactor has been totally 6800 hrs on pressure, mainly 28.5 ato and the integrated power at the end of 1963 was 1340 M W d corresponding to 950 M W d/ton UO_2.

The heavy water has degraded from 99.74 to 99.68% D_2O, and the average D_2O losses in the period between the inventory checks (performed during shut down and at low temperature) have ranged from 100 g/hr to 700 g/hr. Two incidents have resulted in a loss of about 300 kg D_2O. Sampling of the tritium content in moisture samples in the containment ventilation system has gradually developed to be a valuable tool for checking the daily D_2O losses. Tritium sampling of the closed secondary system water has shown no measurable leak-over of D^2O in the heat exchangers.

One fuel element failure has taken place resulting in fission product release. The I 131 content in the D_2O went up a factor 300 above the initial background which was about 10^{-6} μC/cc, but is back to normal after removal of the failed element.

The main flange connection for the reactor vessel has not been touched since installation in 1958, and no measureable leakages have been observed in the aluminium gaskets. Only a smaller fraction of the D_2O losses originates from the 335 seals in the top lid. 140 fuel positions seals have been opened and closed since the high temperature operation started. There has been some equipment failures, however, most of these have in general been of such a nature that they did not cause serious delays. Modification of the gas bleed off from one of the steam transformers has been necessary due to accumulation of radiolytic gases. Tests have proved that the radiolytic gas accumulation in the control station thimbles is not serious from an explosion point of view. This has resulted in that gas bleed off from the control stations could be avoided.

All operations are remotely controlled from the control room, which is located 150 m from the reactor hall. Since the shielding arrangement for the reactor does not permit access to the reactor hall during operation, the instrumentation of the reactor plant is rather extensive. The control of the reactor and plant has caused no significant difficulties. The reactor proved, however, to be more sensitive to load changes than anticipated, as a sudden increase in steam load outtake initiated flash boiling in the moderator.

The activity levels in the reactor hall did not raise any significant operational or maintenance problems. The highest closed contact dose rate measured 15 min. after shut down from 20 MW was 500 mr/hr (primary subcooler). The tritium activity in the D_2O increased to 120 μC/cc during the period.

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Abstract

WATER CHEMISTRY RESEARCH AT HBWR

L. Hammar, R. Rose and G.M. Allison*)
Institutt for Atomenergi, OECD Halden Reactor Project,
Halden, Norway

The effects of various parameters on the net rate of water decomposition in a boiling heavy water reactor were investigated in the HBWR mainly to establish the feasibility of controlling the water chemistry by addition of deuterium and ammonia to the feed water. A further objective was to study the behaviour of the isotope N^{16} in a boiling system. The experiments were carried out at power levels in the range 5 - 18 MW and at pressures from 10 to 28 ata.

Under the conditions investigated, the net rate of water decomposition, which was of the order of 5 - 10 ml O_2 (STP) per kg of steam, i.e. smaller than reported for boiling light water reactors, was attributed mainly to radiolytic processes occuring in the boiling channels. Radiolytic equilibrium was attained in the moderator at rather low concentrations of radiolytic gas (about 0.05 ml O_2 per kg D_2O or less) and stripping of this gas from the moderator on passage through the fuel channels by natural circulation gave a relatively small contribution to the net rate of water decomposition.

An appreciable influence of parameters affecting the gas stripping efficiency in the fuel channels was observed, indicating a significant recombination of oxygen and deuterium formed in primary radiolytic processes. Thus an increase of the specific net rate of water decomposition (formation of radiolytic gas per kg of condensed steam) was observed

*) Paper prepared and presented on behalf of the OECD Halden Reactor Project - a joint undertaking of the European Nuclear Energy Agency.

with increasing steam production (exit steam quality up to 4%), increasing boiling density (i.e. increasing reactor power at constant steam production) and decreasing pressure (28 to 10 ata).

In accordance with the observation that, already without any additions to the feed water, radiolytic recombination occurs to an appreciable extent in the fuel channels, a marked reduction of the net rate of radiolysis was obtained with addition of deuterium to the feed water, e.g. from ca. 6 to ca. 0.03 ml O_2 per kg of steam at 10 MW and 28.5 ata. Similar to the observations with no addition to the feed water, the effect of deuterium addition showed a marked dependence on the stripping conditions in the fuel channels.

By adding deuterium to give about 0.3 - 0.4 ml D_2 per kg of moderator, the decomposition of ammonia added to the moderator into nitrous and nitric acids is prevented. The radiation stability of the ammonia with respect to decomposition into nitrogen and hydrogen was furthermore found to increase with increasing deuterium concentration in the moderator, in agreement with earlier results obtained at room temperature by one of the authors.

Due to the particular design of the primary system, the total concentration of gases which can be maintained in the moderator of the HBWR is rather limited (ca. 5 ml/kg). Because of this limitation, the possibility of producing ammonia in any appreciable concentration in the moderator (e.g.> 0.1 ppm) by radiation induced fixation of added nitrogen and deuterium was not feasible.

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Abstract

THE NPY-PROJECT
A CO-OPERATIVE RESEARCH PROGRAMME IN REACTOR PHYSICS BETWEEN NORWAY-POLAND-YUGOSLAVIA AND IAEA

by

V.O. Eriksen Norway
N. Raišić Yugoslavia
R. Zelazny Poland
B. Semenov IAEA

(Submitted by Institutt for Atomenergi, Norway)

ABSTRACT

It is expected that many advantages could result from a co-ordination of research programmes in reactor physics of different national institutions so as to permit a concerted attack on problems of common interest. The NPY-Project between Norway-Poland-Yugoslavia with IAEA as a catalyser is an attempt to explore the merits of a co-operative effort in reactor physics.

The main objectives of the NPY-Project are that the Signatories should co-operate in the development of experimental techniques in reactor physics and to test and check their applicability, validity and accuracy and they should further develop theoretical models and methods used in lattice calculations, in interpretation of experimental data and test the extent of their validity.

An outline of the joint tasks undertaken will be given.

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Abstract

INTRODUCTION OF ROOPPUR NUCLEAR POWER PLANT IN EPWAPDA GRID

Mohammed Yusuf
Principal Engineer, Pakistan Atomic Energy Commission

The 70 MWe Rooppur Nuclear Power Plant for which construction bids have been invited is scheduled to start commercial operation in the power grid of East Pakistan Water and Power Development Authority (EPWAPDA) in the beginning of 1969. The nuclear reactor of the Plant will be either boiling water or pressurised water type. The EPWAPDA grid is presently under organisation and rapid expansion. Four small grids which are at present working as independent systems are scheduled to be all interconnected by 1969. The total generating capacity of all these small grids at present is 175 MW, whereas the total peak demand on them is 110 MW. This demand is estimated to grow by 1970 to 440 MW because of the extensive development programme currently in progress in East Pakistan, and the total generating capacity planned to be available at that time in the combined grid is 536 MW out of which 460 MW will be firm, so that the 70 MW Rooppur Plant will constitute about 12 per cent of the total generating capacity in the grid when it will start full commercial operation. The cost of imported fuels in East Pakistan varies from 60 to 80 cents per million BTU. The cost of development of her only hydro potential of 80 MW has been extremely high - $1250 per kW installed. Consequently, as this paper shows, Rooppur Plant is estimated to produce power cheaper than any conventional plant including hydro. Only gas-fired plants in certain locations of the Province will be an exception. But the Rooppur Plant will serve as base-load station in the EPWAPDA grid in any case, because its cost of incremental power is estimated to be less than that of power from a gas-fired plant.

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Abstract

A. Leite Garcia, Electrical Engineer,
Companhia Portuguesa de Indústrias Nucleares, Lisbon

J. Cruz Morais, Chief Engineer,
Repartidor Nacional de Cargas, Lisbon

Sidónio Pais, Chief Engineer
Grémio Nacional dos Industriais de Electricidade, Lisbon

The authors describe in a short introduction the present, mainly hydro - electric, power system in Portugal, and then refer to its future development in comparison to the estimated growth of power consumption. A brief evaluation of national energy resources is also made. These subjects are not dealt with in detail.

A first chapter is devoted to analysing the main technical factors on which depends the integration of conventional and nuclear thermal plants within the present power system.

The main factor concerns the way thermal plants may be placed within the load durations. A model is defined to study this factor, which attempts to evaluate the possibilities of concentrating energy production in those thermal plants having the lowest variable costs, without impairing optimum use of hydro incoming energy, and eventually without prejudice to the need for assigning supplementary thermal capacity to peak load duty during periods of high water flow. Application of this model leads to curves showing the future development of the thermal capacities that may be placed within the load durations under a constant marginal criterium. A few remarks are added concerning the operating characteristics of the various levels of thermal capacities, as drawn from the above curves. Those cha

racteristics pertain to the needs for daily and weekly load variation.

The factors determining the size of the largest generating set in operation are then evaluated, this analysis being made in qualitative terms owing to a lack of economic data concerning the portuguese network. Regarding the siting of future thermal plants of various types, geographic and economic aspects are considered in relation to the regional distribution of power consumption.

The second chapter attempts to evaluate the possibilities of integrating nuclear plants within the national power system, having in due regard the technical factors referred to above. Technical and economic characteristics are referred concerning the plants of various types to be considered: hydro plants, conventional thermal plants, thermal plants for peak duty, and nuclear plants. Changes in several basic economic parameters are then taken into consideration, and some possible programs are established for the integration of nuclear plants.

A last chapter indicates briefly the main impacts on the country's economy that should be expected from the adoption of these programs. An evaluation is made of foreseeable trends regarding namely the cost of electric power and the indirect effects on industrial development, on the balance of payments and on the capital market.

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Abstract

PHYSICS AND ENGINEERING DESIGN OPTIMIZATION OF NUCLEAR POWER PLANT USING GRADIENT METHODS

by

Mr. B.M. Nel[1] and Prof. H. Fenech[2]

M.I.T. Computation Centre, Cambridge
Massachusetts

The problem of designing an optimum nuclear power plant amounts to the minimization of some objective function, e.g. the function defining the unit cost of electricity in terms of the relevant design variables, subject to a number of constraints. These constraints include the design equations governing the relationship between these variables, the restrictions on the magnitude of some variables, etc. This nonlinear problem can be solved by appropriately modified gradient techniques such as the method of steepest descent, the optimum gradient method, the accelerated optimum gradient method and Newton's method. The application of these methods to the above mentioned problem are described using a specific nuclear power plant as model.

Gradient methods generally require an initial design which must satisfy all the constraints; this can be a laborious problem. Procedures are described which will automatically produce such a design from an approximate set of values for the design variables.

[1]Research Officer, South African Atomic Energy Board

[2]Associate Professor of Nuclear Engineering, M.I.T.

Two major problems which beset gradient procedures are: 1) errors propagated by the method, and 2) convergence. The first is tied in ultimately with the inversion of matrices of large order. This problem is discussed and illustrated with results obtained for a case problem. Techniques whereby the convergence may be accelerated are described and results obtained by their use given.

The problem of trying to establish whether there are several relative minima in the field of interest to designers are mentioned and a technique described which constrains the gradient methods such that they converge to a saddle point, if such a point exists, from which all the surrounding relative minima may be found.

A simplified block diagram is included showing the assembly of all the above procedures in a form suitable for coding a digital computor program.

The procedures described have been applied to a nuclear power plant involving a total of 63 parameters and 48 constraints and programmed for the IBM 7094 computor. The complete power plant design, including engineering and neutron physics, was considered in order that there may be free interplay between parameters in different sections of the plant, the purpose being to attempt to design an over-all optimum plant rather than an assembly of optimum sections as is often the case. Available results illustrating convergence rates, round-off error in the resulting 48 x 48 matrix, and times for the different operations are given.

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Abstract

THE MECHANISM OF NUCLEATE BOILING

by

Dr. C.J. Rallis[1] and Mr. H.H. Jawurek[2]

Department of Mechanical Engineering, University of the Witwatersrand, Johannesburg

Before the full potential of boiling as a means of heat transfer can be realized it is necessary to have an understanding of its basic mechanism. This paper represents an attempt to approach the problem in a simple and direct manner. Only the simplest case of boiling, that is, nucleate boiling under saturated pool conditions, is dealt with. Treatment of this simple case is considered a prerequisite to any more advanced study of boiling.

For many years latent heat transport, defined by

$$(q/A)_{LH} = \rho_v \lambda_v \sum_{i=1}^{n} f_i V_{di} / A = N \overline{fV}_d \rho_v \lambda_v$$

has been considered to contribute negligibly to the total heat flux under all nucleate boiling conditions. Experimental results on nucleate pool boiling at saturation, reported herein, indicate that the ratio $(q/A)_{LH}/(q/A)_{Tot}$ increases steadily with increasing flux, tending to unity as $(q/A)_{Tot}$ tends towards burnout. Latent heat transport and convection together account for the total flux

[1]Senior Lecturer. [2]South African Atomic Energy Board Bursar

is saturated boiling.

The bulk of this paper deals with the parameters entering into the formulation of $(q/A)_{LH}$. Inevitably these must also have a bearing on any bubble-induced convection mechanism.

The two most important parameters entering into $(q/A)_{LH}$ are (i) the bubble source concentration, N, which is primarily surface-dependent, and (ii) the mean mass flow rate from each source, $\overline{fV}_d\rho_v$, which is essentially hydrodynamic in nature. However, these two parameters are interrelated.

Experimental results show that (i) the arithmetic mean product $\overline{fV}_d$ increases with flux, and (ii) at a particular flux the product fV_d is approximately the same for each bubble source.

The term N, the bubble source concentration, is of prime importance from the point of view of heat transfer correlation, in that it enters both into the formulation for latent heat transport, and into that for convection as the number of liquid-flow inducing sources. In the region of isolated bubbles the bubble source concentration, N, is identical with the nucleation site concentration, $\mathcal{N}$; at higher fluxes this is no longer the case, since vapour from several nuclei can coalesce to form only one bubble. The relation

$$(q/A)_{Tot} = \text{const.}\mathcal{N}^n$$

where $n \simeq 0.5$ is confirmed for water boiling at atmospheric pressure and for organics at various low pressures.

Using this relation and an equation characterizing nucleation from a cavity, and given a boiling curve, it is found possible to arrive at the size distribution of potentially active nucleation cavities; that is, a surface can be characterized with respect to nucleation. Data on both $\mathcal{N}$ and N, and hence coalescence, are presented.

Analysis using the nucleation equation indicates that nucleation and hence nucleate boiling cannot be made to proceed indefinitely at lower and lower pressures; for a given surface-liquid combination a pressure exists below which nucleation cannot proceed in the normal sense. Confirmatory experiments show that this pressure marks the threshold of a hitherto unrecognized regime of unstable nucleate boiling. At lower pressure the nucleate regime is altogether absent.

Whilst still in the early stages of development, the study here reported illustrates the lines along which a rational nucleate boiling heat transfer correlation could be obtained.

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Abstract

THE "DON" PROJECT

F. Pascual, L. Palacios and M.K. Sanders
Junta de Energía Nuclear. Madrid
Atomics International. Canoga Park. Calif.

The DON Project is dedicated to the development of a heavy water-moderated, organic-cooled system which can operate on natural uranium fuel when built on a large scale.

A 30 MWe reactor has been designed to serve as a prototype. Experimental facilities have been incorporated to permit the irradiation of nuclear materials and fuel elements. Isotope production channels have also been included.

A Title I design has been completed by the JEN and the American firm Atomics International. Additional research and development work is also underway where needed.

The cool moderator is contained in an aluminium calandria. The hot coolant circulates from a toroidal inlet header through flexible lines to hot pressure tubes located within the calandria tubes. It removes the heat generated in the fuel elements and flows through flexible lines to the toroidal outlet header. The pressure tubes are made of SAP alloy

Translation provided by courtesy of the submitting Government

joined to steel and pieces.

The fuel element is a 19-rod cluster of UC pellets clad in finned SAP tubes. Control is provided by B_4C rods submerged in the moderator.

Specifications of high-density, arc cast UC are discussed in the paper. SAP quality is considered satisfactory for fuel sheaths; better knowledge of its long term properties is desirable for the process tubes. Preliminary results on UC-SAP compatibility are encouraging. No appreciable reaction is found below 500°C. Compatibility tests are underway both at AI and JEN facilities. These tests point out to hyperstoichiometric UC for SAP-clad elements. Development work is being carried out on SAP-Stainless steel joints, control rod drives fuel handling machine and other areas.

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Abstract

Manufacturing and Installation Experience from the Heavy-Water Systems of the Ågesta Nuclear Power Plant

P.-I. Nylander [1], P. Hjertberg [1], B. Larsson [1]

The Ågesta Nuclear Power Plant (R3/Adam), which is designed for combined district heating and power generation and is located 15 miles south of Stockholm, Sweden, went critical on 17th July, 1963. The pressurized heavy-water reactor of the pressure-vessel type is fuelled with natural uranium.

The paper will present some of the experiences obtained from the design, manufacture and installation of the components in the main cooling and auxiliary heavy-water systems.

The demands for a high degree of tightness, compact structures with small heavy-water volumes and for smooth internal surfaces for the components, mostly made of stainless steel, have necessitated special designs for a number of the components. A brief survey is given in the paper of the design of the components for the reactor portion of the plant and also of the testing performed during the design stage.

The electrical equipment was tested in three successive stages, namely in the testing department of the manufacturer, in the laboratory of the main contractor and, finally, at the site after the erection had been completed. Testing prior to delivery covered practically all the equipment and comprised tests difficult to perform at site. All the cables and cable penetrations at the site were impulse tested or tested with high-voltage d.c. The correct wiring of all circuits was checked against circuit diagrams, and the functional testing of the apparatus, instruments, etc., was repeated.

1) ASEA, Västerås

The paper deals also with the most important questions concerning the design and erection of the pipework in the main cooling system. The choice of materials and the stress analysis are some of the subjects discussed. The erection of the pipework and especially the realization of the cold springing are described. The results were checked by straingauge measurements at ambient and elevated temperatures, and information is given of the application of the strain gauges and the experiences obtained with high-temperature gauges. The measured and estimated stresses are also compared.

Stringent cleanliness requirements were observed during the manufacture and installation so as to ensure good functioning of the plant without too many disturbances during the commissioning, even though the internal cleaning of the systems after erection was to be carried out only by means of the circulation of demineralized water, i.e., without any pickling.

The paper describes how the components were cleaned during manufacture and how the clean erection work at the site was organized.

Operational experience from the plant will be presented in a separate paper at the Conference.

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Abstract

The Design of the Marviken Boiling Heavy Water Reactor with Nuclear Super-Heating

P.H. Margen [1], E. Höjgaard [2], L. Leine [3], R. Nilson [1]

The second Swedish power reactor, Marviken, was originally intended to be a pressurized heavy water power reactor for 105 Mw elect. Studies on larger reactors made during 1961 and 1962, however, suggested that direct cycle boiling heavy water reactors with natural circulation were feasible and that they effected considerable reductions in capital costs due to the entire elimination of the primary cooling circuits with steam generators. It was therefore decided to redesign Marviken as a reactor of this type and to incorporate in addition the possibility of nuclear super-heat which had also been studied in some detail. For the new design the output of Marviken was increased to 140 Mw elect. for saturated steam operation and 200 Mw elect with nuclear super-heat operation. Manufacture of the pressure vessel started in 1963 and the station is scheduled for commissioning in 1968.

The boiling elements consist of Zircaloy clad UO_2 rod fuel clusters whilst the first super-heater charge is planned with Cr/Ni alloy clad UO_2 rod clusters. The average burnup is currently scheduled for 13000 Mwd/t-U and the average enrichment of the entire fuel charge with super-heat operation will be about 1,35 %. Natural uranium operation is possible with saturated steam. The engineering features include on-load fuelling of the elements for the boiling channels. The containment is of the pressure suppression type with a dry well of pre-stressed concrete. The turbine has special measures for minimizing leakage.

1) AB Atomenergi, Stockholm
2) State Power Board, Stockholm
3) ASEA, Västerås

The paper describes the reactor and station and reviews the main development work carried out for Marviken. Reactor operation and safety aspects are also discussed.

The development work includes extensive tests on hydrodynamic instability, static and dynamic burnout, steam water separation, can corrosion and erosion, heat transfer with super-heated steam on roughened surfaces, zero energy physics, water chemistry etc., and calculations on related topics such as void feed back instability and form factors. Leakage prevention for components in the direct cycle has been investigated in a special programme including tests on two 160 Mw elect. condensers in Sweden and tests on a full scale turbine shaft seal. The work completed so far has given considerable encouragement concerning the performance with saturated steam, and the first information concerning performance with super-heated steam is also accumulating.

It is hoped that the information produced by the design, construction and operation of the Marviken station will form the foundation for building a larger fully competitive nuclear power station as the next step.

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Abstract

Current Methods for Long-Term Planning of Electricity Supply in Sweden

N. Holmin [1], G. Lindström [2], M. Mårtensson [3]

The Swedish power system is at present almost entirely based on hydro power but as the cheap hydro resources are being utilized new generating capacity will mainly be thermal in the 1970's.

With special reference to Swedish conditions the methods used by the power companies for optimising their power systems are described.

Different methods used for near-term and long-term forecasts of power consumption are given. The importance for the long-term planning of parameters such as cost of capital and labour, fuel costs and other parameters is discussed. The cost functions for different means of power generation, namely hydro power stations including storage, conventional steam power stations and nuclear power stations, and transmission are given, and the methods used for forecasting possible future trends in relative costs are presented. The problem of reserve capacity as a limiting factor with regard to unit size is discussed.

Based on the forecasts with regard to demand and costs, the most economical composition of the future power generation system is calculated with methods based upon programmes used by the power companies to achieve optimum utilization of water storage resources. A detailed presentation is given on methods used for optimising a system containing generating capacities in the form of conventional steam power and nuclear power with different cost characteristics as a supplement to hydro power. The distinction between a

1) State Power Board, Stockholm
2) AB Skandinaviska Elverk, Stockholm
3) AB Atomenergi, Stockholm

predominantly hydroelectric system and a pure thermal system with regard to the function of thermal power and, specially, nuclear power is discussed. In a hydro electric system the nuclear power has to be matched not only to the conventional thermal power, but also to the water storage reservoirs and the in-flow statistics to those. Thereby, it is possible to achieve a high load factor for the nuclear power, by means of the storage system itself and an operating scheme based on the evaluation of the incremental water value.

Numerical examples are given for Swedish conditions and in particular the economical penalties inflicted, if costs should develop in another direction, different to the one used in optimising the system.

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Abstract

A Large Swedish Nuclear Power Station with a Pressurized Heavy Water Reactor with an Open Uniform Lattice Core.

A. Jonsson [1], B. McHugh [2]

The work presented here covers a study carried out of a nuclear power plant capable of being built today in Sweden using Swedish resources. The major design effort was initiated at the beginning of 1962 in the belief that only by carrying out a design based on realistic conditions would it be possible to raise serious interest among the potential power plant operators.

The plant is designed in accordance with Swedish practice and with full regard to Swedish conditions, and has a net output of 350 MW in a one-reactor two-turbine installation. The general plant design resembles closely a PWR type plant in that it utilizes a pressure vessel reactor with multiple steam generator loops feeding the turbine plant with saturated steam.

It has been found throughout the design of the entire station that the characteristics of the uniform lattice open core give a plant where simplicity of equipment and ease of operation combine with strong inherent safety features to give a low capital cost. The use of D_2O gives the additional advantage of a low enrichment fuel cycle offering the possibilities of operation ranging from natural uranium fuel through the "once through, throw away" U-235 cycle to the U-nat feed plutonium cycle.

The use of fuel rod clusters in guide tubes or pressure tubes has been avoided by distributing the fuel rods in a uniform lattice over the entire core cross-section. This configuration, which leads to an intimate contact between moderator and coolant gives, in conjunction with the volume ratios dictated by the use

1) Axel Johnson Institutet för Industriforskning, Nynäshamn
2) AB Atomenergi, Stockholm

of D_2O, many important advantages when compared with all other water reactor cores.

A virtual absence of hot channel factors combined with the low form factors inherent in the use of D_2O allows the mean thermal loading of the fuel to be set at a value considerably higher than is normal in water cores. The use of an open core with D_2O also means that there is no practical limitation imposed on the core due to burn-out risks. The physics of the core is such that with UO_2 enriched to 1.5 % U-235 in Zr-canning it is possible to achieve a mean discharge burn-up of about 15 000 Mwd /ton U. Operation is possible at a reduced power level with natural UO_2 and in a self-substaining Pu-recycle mode of operation a burn-up of 9 000 Mwd/ton U is possible.

The study has been taken to a point where price tenders have been obtained for all components and installations, giving a realistic background to the judgement that this type of reactor can be a serious competitor for coming nuclear stations to be commissioned in the early 1970's. It is anticipated that future simplification in plant equipment, with consequent improvements in economy, is possible by exploiting to a greater degree than at present the unique characteristics of the uniform lattice open core pressurized heavy water reactor.

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Abstract

Studies of the Frequency and Power Regulation of a Power System Containing Nuclear and Water Power Stations

D. Jungnell [1], T. Spanne [1]

In an electric power system the proportion of nuclear power is growing and consequently it will play a more important role in the frequency control of this system. Owing to the accuracy required in frequency control special demands will be made upon the frequency regulating nuclear power stations.

For this reason the suitability of using pressurized and boiling heavy water reactor plants together with water power plants for frequency regulation in the Swedish network will be studied in this paper. The Swedish network contains to-day a total power of about 8 000 Mw. The fast load variations are normally not over 1 %. In the frequency regulating stations electrical governors with a permanent statism of down to 0.4 % are used. Normally the frequency regulation is chosen so that the permanent change in power production will be a little more than 50 % of the total power at a frequency change of 1 Hz.

Studies of frequency regulation by nuclear power plants will be carried out on an analogue computer, for which reason a dynamic model of the power system is developed. The frequency regulating nuclear power stations och the other frequency regulating stations are treated as two separate parts while the remaining part of the power system is represented by a simplified empiric transfer function between power and frequency. As the compensating contribution of the remaining network to the frequency control is small in comparison to the contribution of the frequency regulation stations the accuracy of this model of the remaining network can be considered sufficient.

1) State Power Board, Stockholm

The reactors of the regulating nuclear power stations are assumed to be of the boiling and pressurized heavy water types. For these stations a detailed analogue model is used, including lumped models of the reactor with control system and the steam turbines with valves and control system, so that transients in power, pressure and temperatures in different parts of the reactor system can be studied.

The conditions at both regulated and unregulated reactor power at different values of the statism of the turbine governors will be studied. Different regulating principles for adjusting the reactor power to the load as well as the influence of transients on the reactor and the main components will be investigated. A comparison between nuclear and water power stations used for frequency regulation will be done and the cooperation between these stations at various load disturbances will be specially considered.

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Abstract

Hydrodynamic Instability and Dynamic Burnout in Natural Circulation Two-Phase Flow. An Experimental and Theoretical Study.

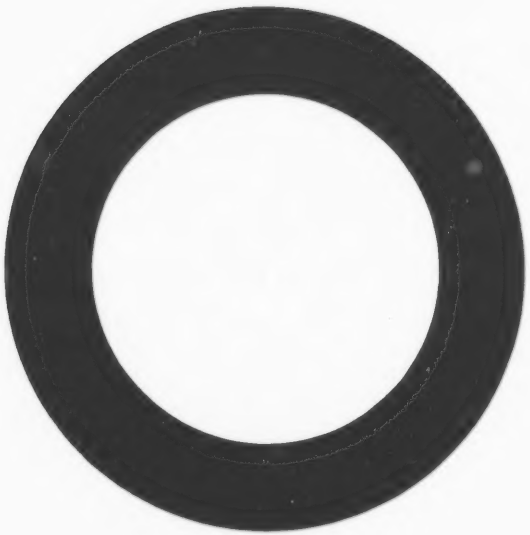
K. M. Becker [1], I. Haga [1], P. T. Hansson [1], R. P. Mathisen [1]

A theoretical model for predicting the threshold of instability for two-phase flow in a natural circulation loop is presented. The model calculates the flow transient caused by a step disturbance of the heat input, and is based upon the conservation laws of mass, momentum and energy in one-dimensional form. Empirical correlations are used in the model for estimating the void fractions and the two-phase flow pressure drops. The equations are solved numerically in a finite difference approximation coded for a digital computer. Two versions of the model exist; one where the total pressure drop over the channel is prescribed and one where the total pressure drop is calculated from the downcomer data.

An experimental study of the hydrodynamic instability and dynamic burnout in two-phase flow has been performed in a natural circulation loop in the pressure range from 10 to 70 atg. The test sections were round ducts of 20, 30, and 36 mm inner diameter and 4890 mm heated length. The experimental results showed that within the ranges tested, the stability of the flow increases with increasing pressure and increasing throttling before the test section, but decreases with increasing inlet sub-cooling and increasing throttling after the test section. Comparing the natural circulation burnout steam qualities with corresponding forced circulation data showed that the former data were low by a factor up to 2.5. However, by applying inlet throttling of the flow the burnout values approached and finally coincided with the forced circulation data.

The present experimental results as well as data available from other sources have been compared with the stability thresholds obtained with the theoretical model. The comparisons included circular, annular and rod cluster geometries, and the agreement between the experimental and theoretical stability limits was very good, the deviations being generally less than 10 per cent.

The application of the experimental and theoretical results on the assessment of boiling heavy water reactor design is discussed. It is shown qualitatively how hydrodynamic instability and burnout affect the optimization of the reactor performance.



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Abstract

Fuel Development for Swedish Heavy Water Reactors

H. Mogard[1], S. Djurle[1], I. Multer[1], H. P. Myers[1], B. Nelson[1], and U. Runfors[1]

Recent development work on uranium oxide fuel elements for use in the Swedish heavy water reactors, i. e. Ågesta (in operation 1964) and Marviken (scheduled for operation 1968), is reviewed. Development trends for the fuels of possible future heavy water reactor projects are indicated.
In comparison with the Ågesta reactor, certain improvements in performance and design are proposed for Marviken which together with progress in fabrication procedure are expected to significantly reduce the fuel costs.
The fuel assembly design is characterized by the adoption of long fuel rods and rod spacers constituting a grid. With a slight enrichment of 1.3 %, a burn up figure of around 13,000 Mwd/t is foreseen for the Zircaloy-canned boiler fuel elements of the Marviken reactor. Fabrication costs are given.
The development programme includes in-pile and out-of-pile experiments on prototype fuel elements as well as investigations on some fundamental properties of the uranium oxide fuel and of the Zircaloy canning material.
In particular, results of investigations on the heat conductivity and the substoichiometric composition of the uranium dioxide are reported as are also experimental investigations on the hydrogen embrittlement of Zircaloy.
Available resources including loop facilities in the Studsvik materials testing reactor R2 are described.
In the last year the development work has been largely concerned with super heat fuel elements for possible use in the Marviken reactor and, in particular, with super heat canning materials of Fe-Cr-Ni alloys.
Some preliminary data are reported.

The fuel development work associated with the long range use of recycled plutonium in the Swedish heavy water reactors is also commented upon.

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Abstract

Processes for the Recovery and Purification of Fission Products from Irradiated Uranium

S. Ahrland [1], K. E. Holmberg [2]

Waste solutions from reprocessing plants may be treated in order to achieve one or more of the following objectives: -

1. Solutions still containing the bulk of the uranium: To separate most of the uranium from the fission products in order to achieve a high concentration factor for the final waste.

2. To separate scientifically or technically valuable and long-lived fission products, such as ^{137}Cs, ^{90}Sr, ^{144}Ce and ^{147}Pm, in a pure form.

3. To separate most of the long-lived nuclides ^{137}Cs and ^{90}Sr in order to considerably shorten the requested storage period of the high activity waste.

Among the various methods conceivable we have so far tried ion exchange using inorganic compounds, and also freezing out. These methods may, of course, also be combined with others, e.g. precipitation.

Inorganic Ion Exchangers

Zirconium phosphate and the hydrous oxides of zirconium and silicium (silica gel) have been judged as the inorganic ion exchangers of greatest promise. Zirconium phosphate is moderately acidic, the hydrous oxides only weakly. In fact the zirconium hydrous oxide is amphoteric, acting mainly as an anion exchanger when pH $<$ 5. The selectivity between ions of different charges generally increases with decreasing acidity. On the other hand, a certain

degree of sorption is reached at a lower pH value, the more acidic the gel which may be important for easily hydrolysed ions. For pH > 8, zirconium phosphate is heavily hydrolysed and thus not useful as an ion exchanger. Silica gel can be used up to a pH = 9, where dissolution becomes appreciable. The workable ranges of pH, as well as other properties of the investigated ion exchangers, thus supplement each other rather well.

With zirconium phosphate, Cs^{+} and Ce^{3+} have been isolated from Sr^{2+} as pure fractions, even in the presence of large amounts of Na^{+}. It may even be possible to isolate Cs^{+} from solutions containing UO_2^{2+} in rather high concentrations.

With silica gel Zr(IV) may be selectively sorbed directly from the strongly acidic waste. This cannot be achieved with zirconium phosphate, which reacts irreversibly with four-valent cations, on account of their great affinity to phosphate ions.

Separation of Uranium from Fission Products by Freezing

On partial freezing of a concentrated solution of uranyl nitrate containing a small amount of fission products, uranyl nitrate and ice crystallize leaving the fission products in the mother liquid. The phases may be separated by centrifugation, leaving the solid somewhat contaminated by mother liquid. By repeated partial freezing it should be possible to purify the uranyl nitrate further. The more important fission products have been tested and do not appear to co-crystallize with uranyl nitrate. The freezing out of uranyl nitrate and ice may also be performed in the presence of moderate concentrations of nitric acid, which reduces the solubility of the uranyl nitrate.

It thus seems physically possible to recover a considerable fraction of uranium in a rather pure form from a concentrated uranyl nitrate solution containing small amounts of fission products by repeated fractional freezing. Simultaneously the high activity waste is concentrated in a reduced liquid volume.